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April 5, 2004

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1 and Unit 2
Docket Numbers 50-413 and 50-414
2003 10CFR50.59 Report

Attached please find a report containing a brief description of changes, tests, and experiments, including a summary of the safety evaluation of each, for Catawba Nuclear Station Units 1 and 2 for the year 2003. This report is being submitted per the provisions of 10 CFR 50.59 (d) (2) and 10 CFR 50.4.

Questions regarding this report should be directed to Kay E. Nicholson at 803.831.3237.

Sincerely,

D. M. Jamil

Attachment

IE47

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Units 1 and 2

2003 10CFR50.59 Report

April 5, 2004

This report consists of a summary of changes, tests, and experiments, including a summary of the safety evaluation of each, for Catawba Nuclear Station, Units 1 and 2, for the year 2003. The entries are organized by the type of activity being evaluated in the following order:

17 Type: Minor Modification **Unit:** 0

Title: Minor Modification CE-61592, Delete unused Resistance Temperature Detectors (RTDs) from the Ice Condenser Temperature Monitoring System and abandon in place the Multi Point Selector Switch Assembly with Indicator Panel (2NFP6070)

Description: Minor Modification CE-61592 will delete unused, non Technical Specification related RTDs, along with the associated wiring and conduit, from the Ice Condenser Temperature Monitoring Subsystem. The RTDs and associated wiring will be deleted back to their respective terminal boxes. The cables from the terminal boxes to 1NFP6070 will be taken to ground in the terminal boxes, and remain terminated at 1NFP6070. 1NFP6070 will be abandoned in place to avoid unnecessary work in lower containment. The equipment to be deleted is comprised of three groups of RTDs identified as Floor Cooling, Wall Panel Mounted and Wear Slab Mounted RTDs.

The Floor Cooling, Wall Panel Mounted and Wear Slab Mounted RTDs, which comprise the non-Technical Specification related RTDs of the Ice Condenser Temperature Monitoring Subsystem, do not serve a safety related function nor interact with any safety related components or equipment or any equipment important to safety. This portion of the Ice Condenser Temperature Monitoring subsystem is no longer utilized by any plant personal, yet maintenance on this part of the system is still required. Removal of this equipment will eliminate the need for maintenance. Technical Specification related RTDs will remain active to provide data to the control room where ice condenser temperatures are monitored for anomalies.

Evaluation: This modification has no effect on any of the accidents analyzed in the UFSAR. There are no unreviewed safety questions associated with this modification. No Technical Specification changes are required. UFSAR changes are required for UFSAR Table 6-127 and UFSAR Figure 6-176.

4 **Type:** Minor Modification **Unit:** 0

Title: Minor Modification CNCE-71445, Document the Suitability of Spare Centrifugal Charging Pump Element HS 4574

Description: Modification CNCE-71445 will document the suitability of a spare centrifugal charging pump element identified as HS 4574, which is the job number assigned by the vendor (HydroAire). The new element will be suitable for use in any of the Catawba centrifugal charging pump applications (lA, lB, 2A, or 2B). New supporting documents are added to the station design documents and some existing documents are revised for consistency and to reflect new information developed. The drawing showing performance data for pump element serial number 49778 was retired since that pump element no longer exists. It was remanufactured and is now identified as HS 4574 with new performance data being issued.

The design of the new pump element is similar to that of previous elements except that a different method of securing the balance drum sleeve in place on the shaft was used and only the suction stage impeller was reused. All the other stage impellers (2 through 11) were from a newly refined investment cast pattern. The performance of the pump in acceptance testing performed on October 12, 2001 met the specification except for the horsepower required and the net positive suction head (NPSH) required. These items were reviewed with power systems and ECCS systems engineering and found acceptable. There was concurrence that the horsepower requirements of the new element will be within the capability of the existing power system design. NPSH required for this pump element is greater than the specification called for. The NPSH required at runout (set at Catawba at 560 gpm) is greater than that listed in Table 6-87 of the UFSAR and that referenced in the Catawba response to NRC Generic Letter 97-04. Therefore, this change will require a 10CFR50.59 evaluation.

Evaluation: The design function of the centrifugal charging pump is to serve as the high head safety injection pump in the Emergency Core Cooling System and to serve as the charging pump in the Chemical and Volume Control System (CVCS). Design requirements and operability requirements are described in UFSAR Section 3.9.3.2 (Safety Related Pump Operability). ECCS functions and requirements are described in Chapter 6 of the UFSAR (various sections in 6.3). Chemical and Volume Control functions are described in Section 9.3.4 of the UFSAR. Section 15.4.6 summarizes accident considerations and evaluation associated with CVCS malfunctions.

Design requirements associated with code requirements and Catawba licensing requirements are addressed in the same manner as the previous charging pump element documented in modification CNCE-70520. Revision 1 of the specification did not change code requirements or operability and reliability requirements. Compliance with the specification is documented in the certification of compliance included with the purchasing documentation.

The performance of the pump in acceptance testing performed October 12, 2001 met the specification except for the horsepower required and the NPSH required. These items were reviewed with power systems and ECCS systems engineering representatives and accepted as is. General office and site power systems representatives concurred that the horsepower requirements of the new element will be acceptable and within the capability of the existing designs.

NPSH required for this pump element is greater than the NPSH called for in the specification. The NPSH required at runout - set at Catawba at 560 gpm is greater than that listed in Table 6-87 of the UFSAR and that referenced in the Catawba response to NRC Generic Letter 97-04.

There is no effect on the frequency of occurrence of accidents evaluated in the UFSAR. Even the accidents summarized in Section 15.4.6 are not affected as the new spare charging pump installed in any position will function the same as the one before it. Should this spare need to be used on-line and this is approved an accompanying operability evaluation will further show that the pump will perform in the same manner. Usage of the spare in an outage will be accompanied by ECCS flow verification testing demonstrating that the pump performs as expected. Therefore, the frequency of occurrence for any accident is not affected.

The new spare, if used in any Catawba application, will perform as intended with the same reliability as any pump before it. Primary control of this is established with the purchase specification used and complied with by the supplying vendor. Two aspects of the performance of the pump purchased did not meet the requirements of the specification. The maximum required horsepower of the pump is 690. This is acceptable as stated by general office and site power systems representatives. Any difference associated with the additional 10 horsepower required will only be minimal with regard to the frequency of occurrence of a malfunction associated with emergency power or the prime mover. The service factor rating of the motor is 690 horsepower and is therefore within the defined service of the motors. The required NPSH at the stated runout conditions is approximately 26 feet. This is above the requirements of the specification. Evaluation of this condition shows that available NPSH is minimal in the condition of ECCS injection at full flow at the lowest FWST level, just before going to cold leg recirculation. The established minimum available NPSH is approximately 80 feet. In the eventuality that swap to recirculation for high and/or medium head injection pumps is delayed or fails, at the lowest FWST level a minimum of approximately 73 feet NPSH will be available to the charging pumps. The pump requiring the most NPSH at 560 gpm is HS 4253 (requiring approximately 34 feet). The subject pump HS 4574 requires approximately 26 feet NPSH at 560 gpm. This margin supports the fact that enough water is provided by the Catawba systems for this spare pump. Given adequate water available, this spare pump element will perform with no increase in the likelihood of occurrence of a malfunction.

Given that the frequency of occurrence of a malfunction is not more than minimally increased as discussed above, it is concluded that there will not be a more than minimal increase in the consequences of an accident. The subject charging pump when installed and called upon will provide its accident duty in the same way that existing operable pumps will, so the consequences of any accident are unchanged.

There are no changes in the consequences of a malfunction associated with this activity.

The activity allows for the use of a replacement pump element which is specifically designed and tested for the service intended. The design of the station is only being changed at a sub-component level with no change at the component level associated with the UFSAR ECCS or CVCS descriptions. Stated again, the ECCS and CVCS systems and components as described in the UFSAR Sections 6.3 and 9.3.4 will not be changed. A detail level change will exist in that a new mechanism for retaining the balance drum sleeve on the shaft is used. Also, NPSH required is affected associated with the use of a new impeller on the second stage. NPSH performance should not be affected by impellers beyond the second stage on this eleven stage pump. The original suction impeller was specifically reused in order to minimize the effect on NPSH performance.

The use of the subject spare pump element does not introduce new or different possibilities of malfunctions. This pump element will be just like the operable pumps currently in use. The use of a new balance drum sleeve retaining design and the new impellers still results in the pump being unchanged as it is described and evaluated in the safety and accident analyses.

The subject pump element will be just like the operable pumps currently in use and there will be no effect on any design basis limit for fission product barriers.

A new design for balance drum sleeve retaining is used. This design is specifically based on evolutionary design efforts associated with the specific history of these pumps. The pump design in general is not changed, so the specific statement regarding only the selection of pump designs proven for ECCS type service (UFSAR Section 6.3.2.2) is accurate. The change in NPSH required performance is described above to the effect that system analysis shows that NPSH available is adequate to support reliable pump performance. The new spare pump was purchased according to an appropriate specification which was developed to facilitate a careful and considered replacement purchase. Other than the horsepower required and the NPSH required, the specification was met as attested to in the certificate of conformance. The methods of evaluation used associated with the horsepower required and NPSH required are unchanged.

No Technical Specification changes are required. Changes to UFSAR Section 6.3.2.2 and UFSAR Table 6-87 are required. A 10CFR50.59 evaluation concluded that this change could be implemented without prior NRC approval.

18 **Type:** Miscellaneous Items **Unit:** 1

Title: Calculation CNS-1552.08-00-0335, Catawba 1 Cycle 15 (C1C15) Reload Core Design, C1C15 Reload Safety Analysis, Revision 0

Description: To ensure that the C1C15 reload core needs no prior NRC review and approval, a 10 CFR 50.59 evaluation is performed in accordance with approved administrative procedures. This evaluation determines if a license amendment request (LAR) is required as a result of changes in the physics parameters predicted for this reload. Changes associated with the fuel assembly hydraulic/mechanical designs were previously evaluated. The UFSAR Chapter 15 analyses were also updated for Catawba Nuclear Station using a separate 10 CFR 50.59 evaluation for the change in fuel design. Station modifications, changes in tests, or changes in procedures during the refueling outage must be addressed in separate 10 CFR 50.59 evaluations.

The C1C15 nuclear reload design analysis (SA physics parameters and maneuvering analysis) were performed with an all rod out (ARO) park position for rodded control cluster assemblies (RCCAs) at 222 steps withdrawn or greater. The deeper RCCA park position is needed to minimize RCCA wear in the core upper internals. The evaluation determined that there is no adverse impact on peaking or physics parameters due to RCCA being parked at 222 steps or greater.

Currently, a proposed license amendment request has been submitted to the NRC that relocates reactor coolant system (RCS) related cycle specific parameter limits from the technical specification to the Core Operating Limits Report (COLR). Specifically, the minimum RCS flow is proposed to be reduced to 388,000 gpm upon NRC approval. This proposed RCS flow will be incorporated into the UFSAR Chapter 15 analyses via this 10 CFR 50.59 evaluation. Operation with the NGF LTAs complies with Technical Specification 4.2.1 and has been shown to be functionally compatible. An exemption for the low-tin Zirlo cladding in the NGF LTAs was previously granted.

Evaluation: A 10 CFR 50.59 evaluation is performed for the Catawba Nuclear Station C1C15 core reload in calculation file CNC-1552.08-00-0335. The impact of any other plant changes made concurrent with the refueling outage is not addressed in this evaluation.

The C1C15 REDSAR, performed in accordance with Nuclear Engineering Division workplace procedure NE-102, "Workplace Procedure for Nuclear Fuel Management", and the C1C15 Reload Safety Evaluation confirm the UFSAR Chapter 15 accident analyses remain bounding with respect to the C1C15 safety analysis reactor physics parameters. The safety analysis reactor physics parameters method is described in topical report DPC-NE-3001-PA. The transition core analyses for the RFA fuel type are performed according to DPC NE-2009-PA "Westinghouse Fuel Transition Report", and WCAP-12945-PA, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the McGuire and Catawba Nuclear Stations". Upon NRC approval of the LAR to relocate cycle specific parameters to the COLR, minimum measure flow will be reduced to 388,000 gpm in the COLR.

The C1C15 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. The C1C15 COLR is prepared in accordance with Technical Specification 5.6.5 and submitted to the NRC in accordance with 10 CFR 50.4. Additionally, applicable sections of technical specifications have been reviewed and no changes specifically related to the operation of the C1C15 core are required. Operation with the NGF LTAs complies with Technical Specification 4.2.1 and has been shown to be functionally compatible.. An exemption for the low-tin Zirlo cladding in the NGF LTAs was previously granted.

Therefore, the C1C15 core reload 10 CFR 50.59 evaluation concludes that no prior NRC approval is necessary.

13 Type: Miscellaneous Items **Unit:** 0

Title: Calculation DPC-1553.26-00-0174, CNC-1553.26-00-0295 Westinghouse NGF Assembly Design

Description: Catawba Unit 1 will be operated with eight Lead Test Assemblies (LTAs) in Cycle 15. These LTAs are of a new design designated as the Next Generation Fuel (NGF) design. These LTAs will operate for two to four cycles beginning with Cycle 15.

Evaluation: Duke Power was required to obtain an exemption to 10 CFR 50.46 and 10 CFR 50.44 to allow use of a zirconium alloy different than the previously approved standard Zirlo and Zircaloy-4. The new zirconium alloy to be used in these LTAs is called "low tin Zirlo" and is sometimes referred to as "optimized" Zirlo. The NRC exemption request for low tin Zirlo was approved per an NRC letter from Robert E. Martin to D.M. Jamil, Subject: Catawba Nuclear Station Unit 1 and 2 Re: Exemption from requirements of 10 CFR 50.44, 50.46 and Part 50 Appendix K (TAC Nos. MB6907 and MB 6908), dated August 4, 2003.

After reviewing the Catawba Technical Specifications, it was determined that the only section that could be affected by this activity was the "Design Features" section, paragraph 4.2.1. This Specification states that each assembly shall consist of a matrix of either Zirlo or Zircaloy fuel rods. Low tin Zirlo is interpreted not to fall into the category of Zirlo, since the alloying constituents fall outside the range for Zirlo. However this specification also has a qualifying statement that reads: "A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting locations". This statement is interpreted to allow the use of LTAs that do not conform to the above mentioned two zirconium based materials.

A 10 CFR 50.59 evaluation determined that this change could be implemented without prior NRC approval. No Technical Specification changes are required. No UFSAR changes are required.

16 Type: Miscellaneous Items **Unit:** 0

Title: Compensatory Action for Auxiliary Building Doors AX505, AX506, AX529, and AX530

Description: Auxiliary Building Doors AX505, AX506, AX529, and AX530 are the doors to the Residual Heat Removal System Heat Exchanger Rooms and the Containment Spray System Heat Exchanger Rooms. In order to perform cleaning or other maintenance of these heat exchangers it may be necessary to open the following doors: Door AX506 (1A Heat Exchanger Room), Door AX505 (1B Heat Exchanger Room), Door AX 529 (2A Heat Exchanger Room), Door AX530 (2B Heat Exchanger Room). These doors serve as an environmental qualification zone barrier. The environmental qualification zone barrier protects the Auxiliary Building elevation 577 general area from a high energy pipe rupture of the Residual Heat Removal System or Containment Spray System piping. The doors also afford protection from radiological concerns once an ECCS suction swap to the containment sump occurs. Therefore compensatory actions are needed to ensure this function is satisfied while the doors are open.

Pipe rupture concerns are addressed in the compensatory action by requiring the Residual Heat Removal System Pump and the Containment Spray System Pumps to be tagged out prior to allowing a door to be held open. This eliminates the concern with a high energy line break. The radiological concerns are addressed by requiring the doors to be closed within ten minutes of a safety injection. Based on the analysis presented in calculation CNC-1223.21-00-0004 (Refueling Water Storage Tank Level Setpoints), the Residual Heat Removal System and the Containment Spray System will be drawing suction from the Refueling Water Storage Tank for approximately ten minutes. After ten minutes, swapover to the containment sump can occur and at that time a radiological concern can occur. Closing these doors before the swapover occurs will ensure that no radiological concern will develop in the Auxiliary Building elevation 577 general area.

Evaluation: The compensatory actions of requiring the Residual Heat Removal Pumps and the Containment Spray Pumps to be tagged out and the doors to be closed within ten minutes of a safety injection assures that no adverse effect to the plant will occur from having the doors to these rooms held open. The compensatory action contains the necessary requirements to ensure these steps are taken. Therefore the compensatory action ensures that the design basis of the doors is maintained. The doors to these rooms are not accident initiators. Holding these doors open under this compensatory action will not increase the frequency of occurrence of an accident. The compensatory action returns the doors to their closed position within the time in which they would be needed. Therefore there is no adverse effect to any plant equipment and the liklihood of occurrence of a malfunction is not changed. No Technical Specification changes are required. No UFSAR changes are required. An evaluation per 10CFR50.59 determined that this change could be made without prior NRC approval.

2 **Type:** Miscellaneous Items **Unit:** 0

Title: Compensatory Actions associated with Corrective Action Program Report Number C-02-03469

Description: Catawba Corrective Action Program (PIP) Serial Number C-02-03469 addresses a problem in which the Safety Injection Header was pressurizing to Cold Leg Accumulator pressure. The problem was caused when valve 2NI165 (Safety Injection Pumps Discharge Header to Cold Leg A Check) was upset by the momentary operation of the 2B Safety Injection Pump during a functional test of its breaker. This caused the Safety Injection Pump discharge header to remain pressurized at approximately 650 psig. An associated manifestation of the problem was that Cold Leg Accumulator 2A was noted to be decreasing in level. Attempts to reseat the check valve were not successful.

Engineering evaluated the ECCS as "Operable but Degraded and compensatory actions were developed to check Residual Heat Removal System pressure and vent the piping as required to relieve this pressure until the problem could be resolved at the next Unit 2 refueling outage which was to begin in early March 2003. Other compensatory actions included monitoring the Residual Heat Removal System for gas entrainment and performing periodic flushes of the Safety Injection System pump cold leg injection header to prevent gas binding of the Safety Injection pumps. On February 28, 2002 just before the refueling outage was to begin it was d

Evaluation: The compensatory actions for the Operable but Degraded evaluation for PIP C-02-3469, Revision 3, ensure ECCS components will operate as described in the UFSAR and as assumed in the accident analysis. The compensatory actions use standard venting and flushing methods to confine and monitor nitrogen gas inleakage from the 2A Cold Leg Accumulator. The compensatory actions ensure that assumptions of the UFSAR and the Safety Analysis are met. No changes to the Technical Specifications or UFSAR are required.

Technical Specification Surveillance Requirement 3.5.2.3 states "Verify ECCS Piping is full of water" once every 31 days. The basis for Technical Specification SR 3.5.2.3 states the following:

"Maintaining the piping from the ECCS pumps full of water by venting the ECCS pump casings and accessible discharge piping high points ensures the system will perform properly, injecting its full capacity into the Reactor Coolant System upon demand. This will also prevent water hammer, pump cavitation and pumping non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an safety injection signal or during shutdown cooling)." The basis states "accessible" high point locations. The operability evaluation for PIP C-02-3469 revision 3 discusses the amount of gas liberated into the Residual Heat Removal System piping and that the vent location and vent frequency are sufficient to contain the unventable gas in the inaccessible high points in the containment header and an unventable location in the Residual Heat Removal System crossover line between A and B Train Residual Heat Removal System discharge headers. Furthermore, the evaluation finds that confining the gas to the inaccessible pipe locations will not invalidate the injection times or flow rates assumed in the accident analysis, will not cause pump cavitation or gas binding and will not induce water hammer forces in excess of water hammer events previously experienced by the system without failure of the piping supports. The condition of operability requires the compensatory actions to maintain Residual Heat Removal System and Safety Injection System operability. The compensatory actions increase the frequency of venting at certain locations, increase the

Residual Heat Removal Pump 2A and 2B testing frequency and established a Safety Injection System flush to containment through valve 2NI423 (Safety Injection System Leakoff Header Isolation). These compensatory actions and revision 3 to the compensatory actions do not conflict with Technical Specification 3.5.2. The compensatory actions ensure the accessible portions of Residual Heat Removal System and Safety Injection System piping are filled. Therefore no Technical Specification changes are required.

The compensatory actions used approved procedures to confine and monitor gas accumulation in the Residual Heat Removal System discharge piping. These actions use venting methods similar to those methods used to verify ECCS piping full of water required by Technical Specification SR 3.5.2.3. The Safety Injection System flush procedure has been evaluated to ensure that the system is not rendered inoperable.

The compensatory actions ensure the gas entrainment in the affected ECCS systems is maintained below levels that may cause malfunction of the ECCS components.

UFSAR accident consequences are not increased by the venting evolutions. ECCS venting is approved and required by the Technical Specification SR 3.5.2.3.

The compensatory actions prevent malfunction of equipment important to safety by ensuring gas entrainment is maintained below the manufacturers recommendations. The consequences of equipment malfunction are not changed by the compensatory action.

No new types of accidents are created by the compensatory actions. The operability evaluation finds that the ECCS systems will perform as designed as long as the limits established in the compensatory actions are maintained.

The compensatory action does not create the possibility of an equipment malfunction with a different result than previously evaluated in the UFSAR. The compensatory actions assure that ECCS pumps are protected from potential gas entrainment. The venting methods are similar to established venting and flushing methods.

11 Type: Miscellaneous Items **Unit:** 0

Title: Response to NRC Generic Letter (GL) 97-04, dated January 5, 1998: Assurance of Sufficient Net Positive Suction Head (NPSH) for Emergency Core Cooling and Containment Heat Removal Pumps (Item 2)

Description: Catawba submitted its response to GL 97-04 on January 5, 1998. Part of that response included a section that identified the required NPSH and the available NPSH. A table was provided summarizing, in part, the numerical responses for the Centrifugal Charging Pumps. These values should be revised as noted:

	Required NPSH (feet)	Available NPSH (feet)
Pump	Injection	Cold Leg Recirc
Centrifugal Charging A	was 74 needs to be 72	RHR pump boost
Centrifugal Charging B	was 74 needs to be 72	RHR pump boost

Evaluation: The original required NPSH information was from pump test data that was typical of these model pumps for the Westinghouse orders. Catawba's replacement pump spare (s/n 52147) was tested in 1983 and provided results on file drawing CNM-1201.05-0471.001. This pump was the 1B pump until 1999 then these impellers were reused on the build of the pump that is currently installed in the 1A position (designated HS 4253). Additionally, a new spare pump element (designated HS 4574) has been received which will have 2001 test results provided in file drawing CNM-1201.05-0548.0001, which is being added to our records by modification CNCE-71445. The more limiting NPSH is from the 1983 test results shown on CNM-1201.05-0471.001.

Additionally, calculation CNC-1223.12-00-0058 has been revised to include this updated required NPSH test information. This calculation revision includes the latest hydraulic modeling efforts provided in input calculation CNC-1223.12-00-0057. The major results of the calculation revisions are to improve the treatment of the cases involving Cold Leg Recirculation, by including a more accurate handling of the effects of Residual Heat Removal System discharge piping losses. It is processed under the rules of 10 CFR 50.59. Catawba NPSH calculations have been revised to provide improved modeling accuracy. The results of the calculations are considered as confirmatory in that the original Westinghouse analysis is basically validated. The later purchased charging pumps continue to meet the stated requirement that manufacturer's test data show that at maximum flow the required NPSH will be less than the stated maximum specified (available).

The change is reflected in the above table revision and the reason for the change is also provided above. Modification CNCE-71445 in addition to issuing new design documents associated with recent new spare pump element HS 4574 also provides marked up UFSAR changes to the NPSH discussion in Section 6.3.2.2 and to Table 6-87. A 10 CFR 50.59 evaluation was performed for modification CNCE-71445 that concluded that that new spare pump element is suitable as a drop in spare for all centrifugal charging pumps at Catawba, that the associated document additions, revisions, and deletions are acceptable under 10 CFR 50.59, and that the overall change to the station) which includes revised results of the NPSH calculation as summarized in the proposed UFSAR change package) is also acceptable under 10 CFR 50.59.

12 **Type:** Miscellaneous Items **Unit:** 0

Title: Response to NRC Generic Letter (GL) 97-04, dated January 5, 1998: Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps (Item 3)

Description: Catawba submitted its response to GL 97-04 on January 5, 1998. Part of that response included a section that specified whether the current design-basis NPSH analysis differs from the most recent analysis reviewed and approved by the NRC for which a safety evaluation was issued.

Our response stated that the current design-basis NPSH analysis is unchanged. This statement should be revised as follows:

Later purchased spare and replacement centrifugal charging pumps have required NPSH test data that shows higher required NPSH than that originally used in the ECCS NPSH evaluation. Adequate margin is retained and the change in the Safety Analysis Report is processed under the rules of 10 CFR 50.59. Catawba NPSH calculations have been revised to provide improved modeling accuracy. The results of the calculations are considered as confirmatory in that the original Westinghouse analysis is basically validated. The later purchased charging pumps continue to meet the stated requirement that manufacturer's test data show that at maximum flow the required NPSH will be less than the stated maximum specified (available).

Evaluation: The change is reflected in the table revision (see item 11 in this report) . The reason for the change is also in item 11. Modification CNCE-71445 in addition to issuing new design documents associated with the new spare pump element HS 4574 also provides marked up UFSAR changes to the NPSH discussion in Section 6.3.2.2 and to Table 6-87. A 10 CFR 50.59 evaluation was performed for modification CNCE-71445 that concluded that new spare pump element is suitable as a drop in spare for all centrifugal charging pumps at Catawba, that the associated document additions, revisions, and deletions are acceptable under 10 CFR 50.59, and that the overall change to the station (which includes revised results of the NPSH calculation as summarized in the proposed UFSAR change package) is also acceptable under 10 CFR 50.59.

19 Type: Miscellaneous Items **Unit:** 0

Title: Technical Specification 3.1.7 Bases Revision

Description: During the investigation Catawba Corrective Action Program (PIP) Serial Number C-02-4129, it became apparent that the bases of Technical Specification 3.1.7 are incorrect with regards to the accuracy of Digital Rod Position Indication (DRPI) when the system is in half-accuracy. A new DRPI display system was installed during 1EOC12 and 2EOC11 refueling outages. When the DRPI system goes to half-accuracy (i.e., either Data A or Data B information is unavailable), the system uses the valid data to determine the position of the control rod(s). In the old Westinghouse system, the processors basically double the valid data. The accuracy was +10/-4 if Data A failed and -10/+4 if Data B failed. The new processors use the last DRPI coil (with valid data) to determine the indicated position of the control rod(s). With the new system, the accuracy is -10/+4 when either Data A or Data B fail. The bases will be revised to show this new accuracy.

In addition, several editorial clarifications will be made to the bases:

In the BACKGROUND potion, these include sever "+" signs to "+/-" signs. Due to the design of the DRPI system, it is obvious that these should be +/= signs.

The surveillance requirement for the technical specification requires DRPI to agree within twelve steps of the demand counters. This requirement is met when either Data A or Data B is available. Both are NOT required for DRPI to be considered fully operable per technical specifications. In the LCO section, the failed coil would affect either Data A or Data B (but not both). Therefore, DRPI would still be considered fully operable. The statement is being revised to say "...either Data A or Data B is operable for each rod..."

In the ACTIONS portion, for clarification purposes, the revision is inserting "(Data A and Data B)." The reasoning is the same as stated in the previous paragraph on the LCO section.

Evaluation: The proposed revision to the bases of Technical Specification 3.1.7 will correct the stated accuracy and make several editorial type corrections. The function and operation of the DRPI system is not affected by making this bases change.

The DRPI system is not an accident initiator. Therefore, this activity has no influence of the frequency of occurrence of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). Additionally, this activity does not create the possibility of an accident of a different type than previously evaluated in the UFSAR.

The DRPI system is not safety related and its operation has no affect on any SSC important to safety. Therefore, this activity has no affect on any SSC important to safety.

This activity will have no affect on any design basis limit for fission product barriers. The proposed activity does not involve a method of evaluation; therefore, there has been no departure from a method of evaluation described in the UFSAR.

No technical specification changes are required. No UFSAR changes are required. A 10 CFR 50.59 evaluation determined that this change could be made without prior NRC approval.

20 **Type:** Nuclear Station Modification **Unit:** 2

Title: Modification CN-21428, Revision 0 Upgrade Operator Interface Units, Replace Obsolete Power Supplies and Change Turbine Runback Rated/Termination of Loss of 50 Percent Electrical Load

Description: The Operator Interface Units (OIUs) for the main turbine control system are obsolete and the supplier, ABB/Bailey no longer provides spare parts. Long-term vendor technical support is not guaranteed because of aging workforce and current support being focused on state-of-the-art equipment. Both the control room and the "maintenance" OIU require replacement. The main turbine video monitor in 1MC-1 is being replaced with a new flat screen LCD monitor.

It has been determined that the main turbine control system power supplies should be replaced every ten years. These power supplies have reached the end of their service life and are obsolete; a modification is required to replace them and CN-21428/0 will accomplish this.

At present, the main turbine control system is not linked to the Operator Aid Computer (OAC) because of incompatible technology between the OAC and the existing OIUs. This incompatibility results in an inability to trend and analyze important turbine system parameters. As part of a long range plan to establish a common data storage platform, standard interface will be provided. This modification provides this interface through both OIUs providing turbine process and alarm data to the OAC for display, transient analysis, and long term data archiving.

Some "soft control functionality" will be added by CN-21428/0 which was not included in CN-11428/0 but for which retrofit is planned for Unit 1. Additionally, the scope of VN011428F, which changed the "Loss of Electrical Load" runback rate and termination point, is included in CN-21428/0. This change will modify the Turbine Runback Rate for a loss of 50 percent load. The runback rate will be changed from 15 percent per minute to 18 percent per minute and the endpoint of the runback will decrease from the existing value of 56 percent to the new value of 48 percent load (indicated by first stage turbine impulse pressure of corresponding control valve reference value). The nominal duration of the runback is unchanged at three minutes. This change is intended to allow clearing the Unit 2 high current alarm ("2B Gen Breaker Overcurrent" annunciator on panel 2AD11) that resulted from the Unit 2 automatic runback on June 4, 2003. This variation notice is in response to Catawba Corrective Action Program (PIP) Serial Number C-93-3350, actual corrective action #18.

Evaluation: The Operator Interface Units (OIUs) for the main turbine control system are obsolete and the supplier, ABB/Bailey no longer provides spare parts. Long-term vendor technical support is not guaranteed because of aging workforce and current support being focused on state-of-the-art equipment. Both the control room and the "maintenance" OIU require replacement. The main turbine video monitor in 1MC-1 is being replaced with a new flat screen LCD monitor.

It has been determined that the main turbine control system power supplies should be replaced every ten years. These power supplies have reached the end of their service life and are obsolete; a modification is required to replace them and CN-21428/0 will accomplish this.

At present, the main turbine control system is not linked to the Operator Aid Computer

(OAC) because of incompatible technology between the OAC and the existing OIUs. This incompatibility results in an inability to trend and analyze important turbine system parameters. As part of a long range plan to establish a common data storage platform, standard interface will be provided. This modification provides this interface through both OIUs providing turbine process and alarm data to the OAC for display, transient analysis, and long term data archiving.

Some "soft control functionality" will be added by CN-21428/0 which was not included in CN-11428/0 but for which retrofit is planned for Unit 1. Additionally, the scope of VN011428F, which changed the "Loss of Electrical Load" runback rate and termination point, is included in CN-21428/0. This change will modify the Turbine Runback Rate for a loss of 50 percent load. The runback rate will be changed from 15 percent per minute to 18 percent per minute and the endpoint of the runback will decrease from the existing value of 56 percent to the new value of 48 percent load (indicated by first stage turbine impulse pressure of corresponding control valve reference value). The nominal duration of the runback is unchanged at three minutes. This change is intended to allow clearing the Unit 2 high current alarm ("2B Gen Breaker Overcurrent" annunciator on panel 2AD11) that resulted from the Unit 2 automatic runback on June 4, 2003. This variation notice is in response to Catawba Corrective Action Program (PIP) Serial Number C-93-3350, actual corrective action #18.

Turbine control will not be affected by failure of the OIUs. Control capability afforded by soft controls is within the existing capabilities of the turbine control system; however, the control will be from a new location. The upgraded equipment is seen as a reliability improvement. Thus, no transients should be imposed on the affected unit as a result of this modification.

With respect to the turbine runback rate changes, the runback rated is more aggressive (18 percent per minute versus 15 percent per minute) and progress to a lower power value (48 percent versus 56 percent) over the same duration. The Turbine Bypass System (TBS) is discussed in the UFSAR 10.4.4 and included both the condenser and the atmospheric dump valves. This setpoint change is well within the design capabilities of the TBS. A reactor trip or turbine trip is not more likely to occur since the revised rate is within the capabilities of the TBS. The TBS keeps the transient (loss of 50 percent electrical load) within UFSAR assumptions. UFSAR 15.2.2, 15.2.3, and 15.2.6 remain bounding. A simulator exercise showed all relevant plant parameters will within normal ranges for the transient. This modification does not affect the calculational framework used for evaluating the behavior or response of the facility of SSC.

No UFSAR described functions are degraded as a result of this modification. No UFSAR, technical specification, or selected licensee commitment changes are required as a result of this modification.

1 **Type:** Procedure Change **Unit:** 0

Title: Procedure IP/0/B/3560/008 "Controller Temperatures"

Description: This change to procedure IP/0/B/3560/008 "Controller Temperatures" will delete equipment that has been abandoned. The primary heat trace controller temperature for controllers reviewed in the monthly inspection will be changed to 75 degrees F +/- 10 degrees F. This revision was made because during cold weather periods in January 2003, certain instrumentation lines froze and made the instruments inoperable. An investigation determined that the heat tracing system controllers had an inaccuracy of 15-18 degrees F. These controllers were set at 50 degrees F. +/- 5 degrees F. With an inaccuracy of 15-18 degrees F., the temperature of the heat traced item could be near freezing with the controller set at 50 degrees F. The decision was made to reset the controllers to 75 degrees F. to ensure that the heat traced items do not approach freezing.

Evaluation: The abandonment of equipment was done per the modification process and evaluated per the 10CFR50.59 evaluation for the mod(s). The procedure change updates the procedure to agree with the current status of the equipment. The controller setpoint change has no effect on accidents analyzed in the UFSAR. No Technical Specification changes are required. No UFSAR changes are required. A 10CFR50.59 evaluation determined that this change could be made without prior NRC approval.

5 **Type:** Procedure Change **Unit:** 0

Title: Procedure OP/0/A/6400/006F Revision 36C "Nuclear Service Water System Flush Procedure," Enclosure 4.22

Description: On May 8, 2003, the Containment Spray Heat Exchanger 1A failed a periodic test used to determine the degree of fouling in the shell side of the heat exchanger. Inspection of the heat exchanger revealed clams and corrosion products which originated in the nuclear service water system piping upstream of the heat exchanger. The B train nuclear service water system piping will be flushed to minimize the likelihood of additional clams and corrosion products fouling the essential header components. Procedure OP/0/A/6400/006F Revision 36C "Nuclear Service Water System Flush Procedure", Enclosure 4.22 "Flushing Containment Spray System Heat Exchanger 1B Nuclear Service Water System Supply Line" has been developed to flush the affected piping. Two different flushes will be performed: one through eight inch diameter valve 1RNF80, which flushes the nuclear service water supply to the containment spray 1B heat exchanger; and another flush through valve 1RNE27 which flushes the B train nuclear service water system main supply header. Flushing of this piping will require passing enough flow so that operability of the essential header components cannot be maintained. This could adversely affect the nuclear service water flow rate to the following essential nuclear service water system loads: Component Cooling Heat Exchanger 1B, Auxiliary Feedwater System Train 1B assured makeup, Diesel Engine 1B Jacket Water Heat Exchanger, and the Control Room Ventilation System Chiller B during Safety Injection or Loss of Offsite Power events. The procedure enclosure will ensure that the nuclear service water flush will be secured in a timely manner if necessary.

In order to evaluate the procedure enclosure, the accident flow paths for the 1B nuclear service water essential header were examined. The 1B nuclear service water system essential header is flow balanced to provide flow to the essential loads described above. Each of these loads were evaluated since the nuclear service water supply to the containment spray heat exchanger 1B could be flushed with a flow rate that would affect the operability of the essential header components. Additionally the flush would be secured prior to closing doors (through which temporary flush piping passes) in response to a local tornado watch or warning, or in the event of an Auxiliary Building fire. The flush would be secured if the temporary flush piping breaks or develops a significant leak, or if there was a significant rainfall event at the site and the input into the yard drains exceeded the system capability. The flush would be secured if any of the operators stationed for the flush need to leave their dedicated location for any reason.

Evaluation: The evaluation determined that the potential flow rate problems described above would necessitate quickly securing the flush in the event of a LOOP or LOCA. Either flush would be secured within ten minutes. Securing either flush can be accomplished by closing a single valve. In an accident there would be several minutes before there would be a significant increase in Nuclear Service Water System heat loads. Also provisions were made for quickly securing the flush in the event of a flush piping break, a local tornado watch or warning, a significant local rainfall event, or an Auxiliary Building fire. The procedure enclosure and the guidance contained in it do not significantly increase the risk of accidents analyzed in the UFSAR. No Technical Specification changes or UFSAR changes are required. A 10CFR50.59 evaluation determined that this procedure enclosure could be implemented without prior NRC approval.

6 **Type:** Procedure Change **Unit:** 0

Title: Procedure OP/0/A/6400/006F Revision 36D "Nuclear Service Water System Flush Procedure," Enclosure 4.23

Description: On May 8, 2003, the Containment Spray Heat Exchanger 1A failed a periodic test used to determine the degree of fouling in the shell side of the heat exchanger. Inspection of the heat exchanger revealed clams and corrosion products which originated in the nuclear service water system piping upstream of the heat exchanger. The A train nuclear service water system piping will be flushed to minimize the likelihood of additional clams and corrosion products fouling the essential header components. Procedure OP/0/A/6400/006F Revision 36D "Nuclear Service Water System Flush Procedure", Enclosure 4.23 "Flushing Containment Spray System Heat Exchanger 1A Nuclear Service Water System Supply Line" has been developed to flush the affected piping. Two different flushes will be performed - one through eight inch diameter valve 1RNF79, which flushes the nuclear service water supply to the containment spray 1A heat exchanger; and another flush through valve 1RNE27 which flushes the A train nuclear service water system main supply header. Flushing of this piping will require passing enough flow so that operability of the essential header components cannot be maintained. This could adversely affect the nuclear service water flow rate to the following essential nuclear service water system loads: Component Cooling Heat Exchanger 1A, Auxiliary Feedwater System Train 1A assured makeup, Diesel Engine 1A Jacket Water Heat Exchanger, and the Control Room Ventilation System Chiller A during Safety Injection or Loss of Offsite Power events. The procedure enclosure will ensure that the nuclear service water flush will be secured in a timely manner if necessary.

In order to evaluate the procedure enclosure, the accident flow paths for the 1A nuclear service water essential header were examined. The 1A nuclear service water system essential header is flow balanced to provide flow to the essential loads described above. Each of these loads were evaluated since the nuclear service water supply to the containment spray heat exchanger 1A could be flushed with a flow rate that would affect the operability of the essential header components. Additionally the flush would be secured prior to closing doors (through which temporary flush piping passes) in response to a local tornado watch or warning, or in the event of an Auxiliary Building fire. The flush would be secured if the temporary flush piping breaks or develops a significant leak, or if there was a significant rainfall event at the site and the input into the yard drains exceeded the system capability. The flush would be secured if any of the operators stationed for the flush need to leave their dedicated location for any reason.

Evaluation: The evaluation determined that the potential flow rate problems described above would necessitate quickly securing the flush in the event of a LOOP or LOCA. The flush would be secured within ten minutes. Securing the flush can be accomplished by closing a single valve. In an accident there would be several minutes before there would be a significant increase in Nuclear Service Water System heat loads. Also provisions were made for quickly securing the flush in the event of a flush piping break, a local tornado watch or warning, a significant local rainfall event, or an Auxiliary Building fire. The procedure enclosure and the guidance contained in it do not significantly increase the risk of accidents analyzed in the UFSAR. No Technical Specification changes or UFSAR changes are required. A 10CFR50.59 evaluation determined that this procedure enclosure could be implemented without prior NRC approval.

8 Type: Procedure Change **Unit:** 0

Title: Procedure OP/0/A/6400/006F Revision 37, Enclosure 4.20 and 4.21, Flushing the A(B) Train Nuclear Service Water System Supply Piping Using Crossover Header Flush Line

Description: On May 8, 2003, the Containment Spray Heat Exchanger 1A failed a periodic test used to determine the degree of fouling in the shell side of the heat exchanger. Inspection of the heat exchanger revealed clams and corrosion products which originated in the Nuclear Service Water System piping upstream of the heat exchanger. The Train A and Train B piping will be flushed to minimize the likelihood of additional clams and corrosion products fouling the components supplied by the essential header Procedure OP/0/A/6400/006F R37 Enclosures 4.20 and 4.21, "Flushing the A(B) Train Nuclear Service Water System Supply Piping Using Crossover Header Flush Line" have been developed to flush the A or B Train Main Supply Headers. The flushing of either of these nuclear service water system piping sections willl require passing enough flow so that operability of the essentail header components cannot be assurred while the flush valve is fully open. This could adversely affect the Nuclear Service Water flow rate to the following Nuclear Service Water System loads: Component Cooling Heat Exchangers, Auxiliary Feedwater System assured makeup, Diesel Engine Jacket Water Heat Exchangers, Containment Spray Heat Exchangers, and the Control Room Ventilation System Chillers during Safety Injection or Loss of Offsite Power events. The procedure enclosure will ensure that the nuclear service water flush will be secured in a timely manner if necessary.

In order to evaluate the procedure enclosures, the accident flow paths for the nuclear service Water Essential headers were examined. The nuclear service water system essential headers are flow balanced to provide flow to the essential loads described above. Each of these loads was evaluated to determine if the new flush procedure enclosures can be implemented without prior NRC approval.

Evaluation: AThe evaluation determined that the potential flow rate problems described above would necessitate quickly securing the flush in the event of a LOOP or LOCA. The flush would be secured within ten minutes. Securing the flush can be accomplished by closing a single valve. In an accident there would be several minutes before there would be a significant increase in Nuclear Service Water System heat loads. Also provisions were made for quickly securing the flush in the event of a flush piping break, a local tornado watch or warning, a significant local rainfall event, or an Auxiliary Building fire. The procedure enclosure and the guidance contained in it do not significantly increase the risk of accidents analyzed in the UFSAR. No Technical Specification changes or UFSAR changes are required. 10 CFR 50.59 evaluation determined that these procedure enclosures could be implemented without prior NRC approval.

10 Type: Procedure Change **Unit:** 0

Title: Procedure OP/0/A/6400/006F to allow air sparging of Containment Spray Heat Exchnager 1A during flush, Enclosure 4.21

Description: The Operations procedure for flushing the Containment Spray System Heat Exchanger is being revised to allow air sparging coincident with the flushing process. This change applies only to the Containment Spray System Heat Exchanger 1A. Flushing will allow any residual acid and or by-products from the GOTAR TM chemical cleaning evolution to become suspended in solution. These will be subsequently swept away by the flushing process. Restrictions on maximum acceptable flushing flow have been made to accommodate the possibility of the loss of the sparging connection. This limit on maximum Nuclear Service Water System flow allows operability to be maintained if the sparging connection breaks. Any flooding concerns due to a broken sparging connection are addressed under a separate 10CFR50.59 evaluation [See 10CFR50.59 evaluation for procedure OP/0/A/6400/006F, enclosure 4.20 for nuclear service water inlet pipe flush to Containment Spray Heat Exchanger 1A. (Record number 8 in this report)].

Evaluation: This procedure revision will allow flushing and air sparging of the Containment Spray System Heat Exchanger. Sparging air will be introduced through a Containment Spray Heat Exchanger shell side inlet drain to ensure that residual cleaning by-products are removed by the flushing process. Sparging will provide additional agitation to loose by-products and facilitate by-product removal. This is considered a maintenance activity. It will not jeopardize the ability of the Nuclear Sevice Water System to perform its function. The procedure change restricts Nuclear Service Water System flow to a value which will ensure that the 1A Train of the Nuclear Service Water System remains operable in the event that the sparging connection is lost. A 10CFR50.59 evaluation concluded that this procedure change could be implemented without prior NRC approval. No UFSAR changes are required. No Technical Specification changes are required.

3 **Type:** Procedure Change **Unit:** 0

Title: Procedure OP/0/A/6400/6400/006F, Revision 36A, "Nuclear Service Water System Flush Procedure," Enclosure 4.20

Description: On May 8, 2003, the Containment Spray Heat Exchanger 1A failed a periodic test used to determine the degree of fouling in the shell side of the heat exchanger. Inspection of the heat exchanger revealed clams and corrosion products which originated in the Nuclear Service Water System piping upstream of the heat exchanger. This piping will be flushed to minimize the likelihood of additional clams and corrosion products fouling the heat exchanger when it is placed back in service after cleaning. The piping will be flushed through temporary piping routed through the Auxiliary Building and through doors at the rear of the Auxiliary Building.

A new enclosure for Procedure OP/0/A/6400/006F Revision 36A "Nuclear Service Water System Flush Procedure" was developed. Procedure Enclosure 4.20, entitled "Flushing Containment Spray Heat Exchanger 1A Nuclear Service Water Supply Line", addresses flushing the piping upstream of the heat exchanger. Flushing of this piping will require passing flow in excess of the flow balance flow rate to this component. This could adversely affect the nuclear service water flow rate to the following essential Nuclear Service Water System loads: Component Cooling Heat Exchanger 1A, Auxiliary Feedwater System Train 1A assured makeup, Diesel Engine 1A Jacket Water Heat Exchanger, and the Control Room Ventilation System Chiller A during Safety Injection or Loss of Offsite Power events. The procedure enclosure will ensure that the nuclear service water flush will be secured in a timely manner if necessary.

In order to evaluate the procedure enclosure, the accident flow paths for the 1A nuclear service water essential header were examined. The 1A nuclear service water system wssential header is flow balanced to provide flow to the essential loads described above. Each of these loads was evaluated since the Nuclear Service Water Supply to the Containment Spray Heat Exchanger 1A could be flushed with a flow rate that exceeds the flow balance flow rate. This could have an adverse effect on the Nuclear Service Water flow rate to the essential loads. Additionally the flush would be secured prior to closing doors in response to a local tornado watch or warning, or in the event of an Auxiliary Building fire. The flush would be secured if the temporary flush piping breaks or develops a significant leak, or if there was a significant rainfall event at the site and the input into the yard drains exceeded the system capability. The flush would be secured if any of the operators stationed for the flush need to leave their dedicated location for any reason.

Evaluation: The evaluation determined that the potential flow rate problems described above would necessitate quickly securing the flush in the event of a LOOP or LOCA. The flush would be secured within ten minutes. Securing the flush can be accomplished by closing a single valve. In an accident there would be several minutes before there would be a significant increase in Nuclear Service Water System heat loads. Also provisions were made for quickly securing the flush in the event of a flush piping break, a local tornado watch or warning, a significant local rainfall event, or an Auxiliary Building fire. The procedure enclosure and the guidance contained in it do not significantly increase the risk of accidents analyzed in the UFSAR. No Technical Specification changes or UFSAR changes are required. A 10CFR50.59 evaluation determined that this procedure enclosure could be implemented without prior NRC approval.

9 **Type:** UFSAR Change **Unit:** 0

Title: Revision to Selected Licensee Commitment (SLC) 16.7-10 Revision 0 as related to Radiation Monitor EMF42

Description: Selected Licensee Commitment (SLC) 16.7-10 Revision 0 "Radiation Monitoring for Plant Operations" was revised as follows:

Required Action F.2 was revised from
"Suspend all operations involving fuel movement in the fuel building"
to
"Suspend all operations involving recently irradiated fuel in the fuel building"

Table 16.7-10-1, Item 2, Testing Requirements, was changed from "TR 16.7-10-2" to "TR 16.7-10-2(d)".

A footnote (d) was added which states " The frequency for this CHANNEL OPERATIONAL TEST is 18 months AND within 60 days prior to activities involving recently irradiated fuel."

Two references were added related to Catawba Operating License Amendment 198/191.

Evaluation: Catawba Operating License Amendments 198(Unit 1) and 191(Unit 2) revised Technical Specification 3.7.13 "Fuel Handling Ventilation Exhaust System", to require the Fuel Building Ventilation System to be operable only during the movement of recently irradiated fuel assemblies. The change to SLC 16.7-10 affects radiation monitor EMF42 "Fuel Pool Storage Area - High Gaseous Radioactivity Monitor". This monitor is required to be operable and the testing frequency is required to maintain/demonstrate operability. License Amendments 198 and 191 previously determined the acceptability of the Fuel Building Ventilation System being inoperable during a UFSAR Chapter 15 Fuel Handling Accident, within the restrictions on movement of recently irradiated fuel. Therefore relaxing restrictions associated with the operation of radiation monitor EMF42 to be equivalent with the Technical Specification limitations on the Fuel Building Ventilation System will have no effect on accidents evaluated in the UFSAR. This radiation monitor is not nuclear safety related and does not input to the Reactor Protection System. No Technical Specification changes are required. A UFSAR change to Section 16.7-10 is required. A 10CFR50.59 evaluation determined that this change could be implemented without prior NRC approval.

15 Type: UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment (SLC) addressing the Auxiliary Feedwater Flow Control Valve Air Accumulators.

Description: A new SLiC addressing the Auxiliary Feedwater Flow Control Valve Air Accumulators has been developed. The purpose of the SLC is to define the requirements for operability and testing of the Auxiliary Feedwater flow control valve air accumulators. With the incorporation of this Selected Licensee Commitment, Catawba Nuclear Station will commit that the accumulators will be operable in Modes 1,2,3 and 4 (when a steam generator is relied upon for heat removal). The Auxiliary Feedwater flow control valves are air operated fail-open valves Air pressure is required on the actuators to close the valves. The Auxiliary Feedwater flow control valve air accumulators are designed to provide a minimum of one hour backup air supply to the flow control valve actuators in the event that the normal instrument air supply is not available. The accumulators were installed by modification on both Catawba units. The accumulators are provided specifically for the steam generator tube rupture design basis event - which includes a loss of offsite power and subsequent loss of instrument air. The accumulators are provided to ensure that control room operators can stop flow from the Auxiliary Feedwater System to the ruptured steam generator in the event that the associated auxiliary feedwater motor operated valve is not available for isolation. Several failures have been identified at Catawba that would result in loss of the ability to close an auxiliary feedweter system motor operated valve to a ruptured steam generator. Therefore the auxiliary feedwater flow control valves and air accumulators provide a backup method of steam generator isolation. The ability to isolate a ruptured steam generator allows prevention of steam generator overfill. Prevention of steam generator overfill is a non safety related licensing basis commitment for Catawba Nuclear Station (CNS SER Section 7.3.2.5). The one hour backup air supply provides a temporary method to provide steam generator isolation while steps are taken to manually isolate the affected flow path. Performance of the accumulator leakage test verifies the one hour air supply and thus the operability of the accumulators.

The purpose of the new SLC is to define the requirements for operability and testing of the Auxiliary Feedwater flow control valve air accumulators. The conditions, required actions and completion times were developed based on a severe accident analysis entitled "Risk Analysis for Determining Allowed Out of Service Time for Auxiliary Feedwater Throttle Valve Accumulators."

Evaluation: The SLC is administrative and will not change the configuration or operation of any system, structure or component. There will be no increase in the probability of an accident or malfunction as described in the UFSAR or otherwise. The design basis limit for fission product barriers will not be affected. There will be no departure from any method of evaluation as a result of the implementation of this SLC. No Technical Specification changes are required. A change will be made to UFSAR Section 10.4.9.2 to point out the non-safety, backup steam generator isolation function of the accumulators. An evaluation per 10 CFR 50.59 determined that this change could be implemented without prior NRC approval.

7 **Type:** UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment (SLC), "Auxiliary Feedwater System Flow Control Valve Air Accumulators", is being added to the Catawba SLC Manual. A change to UFSAR Section 10.4.9.2 will be made as well.

Description: A new SLC, "Auxiliary Feedwater System Flow Control Valve Air Accumulators", is being added to the Catawba Selected Licensee Commitments Manual. A change to UFSAR Section 10.4.9.2 will be made as well.
The Auxiliary Feedwater System (AFW) flow control valves (1/2CA36, 40, 44, 48, 52, 56, 60 and 64) are air operated fail-open valves. Air pressure is required on the actuator to close the valves. The AFW Flow Control Valve Air Accumulators are designed to provide a minimum of one hour of backup air supply to the flow control valve actuators in the event that the normal instrument air supply is not available. Station Modifications CN-11391 and CN-21391 installed the accumulators on Unit 1 and Unit 2, respectively. The accumulators are provided specifically for the steam generator tube rupture design basis event - which includes loss of offsite power and the subsequent loss of instrument air. More specifically, the accumulators are provided to ensure that control room operators can stop flow from the AFW System to the ruptured steam generator in the event that the associated AFW motor operated valve is not available for isolation. Several failures were identified that would result in loss of the ability to close an AFW motor operated valve to a ruptured steam generator. Thus, the AFW flow control valves and associated accumulators provide a backup method of steam generator isolation. The ability to isolate a ruptured steam generator affords the prevention of steam generator overfill. Prevention of steam generator overfill is a non-safety related licensing basis commitment for Catawba - as recognized by the NRC in the Catawba Safety Evaluation Report (7.3.2.5). The one-hour backup air supply provides a temporary method to provide steam generator isolation while steps are taken to manually isolate the affected flow path.

The purpose of the new SLC is to define the requirements for operability and testing of the Auxiliary Feedwater flow control valve air accumulators. The conditions, required actions and completion times provided in the SLC were developed based on a Severe Accident Analysis entitled "Risk Analysis for determining allowed out of service time for Auxiliary Feedwater System Throttle Valve Accumulators". The SLC will commit that the accumulators will be operable in Modes 1, 2, 3 and 4 (when a steam generator is relied upon for heat removal).

The SLC has two remedial actions. First, if one or more accumulator is inoperable, then the accumulator(s) must be restored to operable status within a maximum completion time of seven days. Probabilistic Risk Analysis determined that having more than one accumulator inoperable is no worse than only one accumulator inoperable. Second, in the event that an accumulator(s) cannot be returned to operable status within the 7 day completion time, an indefinite completion time may be allowed if specific requirements are met.

1. Both AFW motor-driven pump trains and all associated emergency diesel generator, component cooling water, and nuclear service water support systems must be verified and maintained operable while the accumulator(s) are out of service. The operability of these systems must be re-confirmed once every 12 hours.

2. Action must be immediately initiated to restore the accumulator(s) to operable status. Otherwise, the AFW train(s) associated with the inoperable accumulator(s) must be

immediately declared inoperable and the applicable conditions and required actions of Technical Specification 3.7.5 for the AFW train(s) entered. Probabilistic risk analysis determined that the increase in core damage frequency for an inoperable accumulator(s) is acceptable if operability of the AFW motor-driven pump trains and their associated support systems is maintained. Loss of the turbine driven AFW pump is a possible result of steam generator overfill. Maintaining operability of the AFW motor driven pump trains minimizes the probability that the turbine driven pump will be required to supply flow to the steam generators.

The accumulators were sized assuming a minimum pressure of 80 psig. Thus, the minimum pressure required for accumulator operability is 80 psig. A surveillance requirement for verification of accumulator pressure is not required as an alternate verification method exists. The accumulator system design dictates that accumulator pressure follows instrument air pressure - normally approximately 90 psig. On decreasing instrument air pressure, check valves will close to isolate the accumulators. However, the positioner for each flow control valve consumes air at a constant rate such that accumulator pressure eventually drops to match instrument air pressure. A control room annunciator is provided to notify Operations of a low instrument air pressure condition. The annunciator response procedure and loss of Instrument Air Abnormal Procedure provide adequate guidance for a drop in instrument air pressure, with respect to accumulator operability.

The ability to control Steam Generator (S/G) level and isolate the faulted S/G following SGTR is related to the prevention of S/G overfill. Concerns associated with S/G overfill and recommendations for heightened awareness among operators of the need to prevent it were identified in Generic Letter 81-16. Following the SGTR at the Ginna Nuclear Plant in January, 1982 (following which a S/G was filled), Westinghouse conducted an analysis of SGTR to ensure that the operators could act in time to prevent overfill of the ruptured S/G. The NRC required that each member of the SGTR Subgroup and each Westinghouse near-term operating license submit plant specific information in order apply the methodology or results of WCAP-10698 on a plant specific basis. Each applicable utility was required to submit "a list of systems, components, and instrumentation credited for accident mitigation in the SGTR EOP(s)." Quoting further, "For non-safety grade systems and components state whether safety grade backups are available which can be expected to function ...or justify that non-safety components can be utilized for the design basis event." Each utility was required to perform "a survey of plant primary and "balance-of-plant" systems design to determine the compatibility with bounding plant analysis in WCAP-10698." In addition, "The worst single failure should be identified and the effect of the difference on the margin of overfill should be provided." In its response, Duke reported that "failure of an AFW valve on the ruptured steam generator to close could be mitigated by closing its companion isolation valve (in place of stopping an Auxiliary Feedwater Motor Driven Pump - or Turbine Driven Pump)." In summary, it was noted that "the generic analysis of WCAP-10698 is generally applicable to Catawba." This evaluation was accepted by the NRC. In addition, the NRC acknowledged that the AFW flow control valves were non safety related, backup isolation for the safety-related EMO isolation valves.

Evaluation: An evaluation per 10 CFR 50.59 determined that implementation of the new SLC, "AFW (Auxiliary Feedwater) Flow Control Valve Air Accumulators", could proceed without a license amendment. The purpose of the new SLC is to define the requirements for operability and testing of the Auxiliary Feedwater flow control valve air accumulators. The SLC will commit that the accumulators will be operable in Modes 1, 2, 3 and 4 (when a steam generator is relied upon for heat removal). The SLC is administrative in nature and will not change the configuration or operation of any system, structure or component. There will be no increase in the probability or consequences of an accident or malfunction - as described in the UFSAR, or otherwise. The design basis limits for fission product barriers will not be affected. There will not be a departure from any method of evaluation as a result of implementation of the SLC.

The associated 10 CFR 50.59 evaluation also facilitates an editorial change to the UFSAR Section 10.4.9.2. The change will point out the non-safety, backup steam generator isolation function of the accumulators. No Technical Specification changes are required.

21 **Type:** UFSAR Change **Unit:** 0

Title: Selected Licensee Commitment 16.9-7, Boration Systems Flow Paths - Shutdown

Description: Pursuant to Catawba Corrective Action Program (PIP) Serial Number C-02-2960, this revision proposes to change Selected Licensee Commitment (SLC) 16.9-7 to allow one safety injection pump to be used as the Boron injection flow path in MODES 5 and 6 without the reactor vessel head removed.

The safety injection pump develops sufficient head to borate the reactor coolant system at the Low Temperature Overpressure Protection (LTOP) setpoint and below. The other area of concern is low temperature overpressure protection. The analyses with regards to LTOP is unaffected by this change. As stated in the bases to Technical Specification 3.4.12, the analyses demonstrate "that either one Power Operated Relief Valve (PORV) or the depressurized reactor coolant system and the reactor coolant system vent can maintain the reactor coolant system pressure below limits when only one charging pump or one safety injection pump is actuated. Thus, the LCO allows only one charging pump or one safety injection pump OPERABLE during the LTOP MODES." During the times that the safety injection pump option is used for the boration flow path, compliance with Technical Specification 3.4.12 is required and all other safety injection and charging pumps will be required to be incapable of injecting into the reactor coolant system.

Evaluation: This evaluation applied to a revision of SLC 16.9-7 to allow a safety injection pump to be used as a boron injection flow path in MODES 5 and 6 with the reactor vessel head installed.

No evaluation methods are affected.

The safety injection pump develops sufficient head to borate the reactor coolant system at the Low Temperature Overpressure Protection (LTOP) setpoint and below. The following is contained in the basis for SLC 16.9-9:

"The boration capability of one charging pump, residual heat removal pump, or safety injection pump in association with a flow path and borated water source, is sufficient to provide a SHUTDOWN MARGIN of 1.3 percent $\Delta k/k$ after xenon decay and cooldown to 200 degree F and of 1 percent $\Delta k/k$ after xenon decay and cooldown from 200 degree F to 68 degree F."

The other area of concern is low temperature overpressure protection. The analyses with regards to LTOP is unaffected by this change. As stated in the bases to Technical Specification 3.4.12, the analyses demonstrate "that either one Power Operated Relief Valve (PORV) or the depressurized reactor coolant system and the reactor coolant system vent can maintain the reactor coolant system pressure below limits when only one charging pump or one safety injection pump is actuated. Thus, the LCO allows only one charging pump or one safety injection pump OPERABLE during the LTOP MODES." During the times that the safety injection pump option is used for the boration flow path, compliance with Technical Specification 3.4.12 is required and all other safety injection and charging pumps will be required to be incapable of injecting into the reactor coolant system. The following is contained in the basis for SLC 16.9-9:

"When the temperature of one or more reactor coolant system cold legs drops below 285 degree F in MODE 4, the potential for low temperature over pressurization of the reactor vessel makes it necessary to render all but one charging pump or safety injection pump inoperable. The Technical Specification 3.4.12 limitation for a maximum of one charging or safety injection pump to be OPERABLE and the associated Surveillance Requirement to verify a maximum of one charging pump or one safety injection pump is capable of injecting into the reactor coolant system below 285 degree F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV."

Using a safety injection pump as boration flow path in MODE 5 and in MODE 6 with the vessel head installed is acceptable without an increase in the likelihood or consequences of a malfunction of any equipment. There is no impact to any accidents or malfunctions previously evaluated in the UFSAR. Using a safety injection pump with the vessel head installed is also not seen to present itself as an initiator of any new malfunctions or accidents.

No changes to technical specifications are required. A 10 CFR 50.59 evaluation determined that this change could be made without prior NRC approval.

14 Type: UFSAR Change **Unit:** 0

Title: UFSAR Change to UFSAR Table 7-3 "Reactor Trip System Instrumentation" for Hot Leg RTD Response Time Increase from 6.5 to 8.0 seconds

Description: UFSAR Table 7-3 "Reactor Trip System Instrumentation" indicates the performance characteristics assumed for Reactor Coolant System hot leg narrow range RTDs. The response time will be increased from 6.5 seconds to 8.0 seconds. This change is applicable for Catawba Unit 1 only and is associated with Unit 1 because of the Replacement Steam Generators. Following replacement of the Unit 1 hot leg narrow range RTD's, the A2 RTD was measured to be outside of the current time constant listed in UFSAR Table 7-3. The Nuclear Engineering Safety Analysis Group has documented supporting analyses for an increased time constant that will allow for continued and unlimited future operation with slower time response than currently allowed in the UFSAR.

Evaluation: The slower RTD response time has been fully evaluated to meet applicable accident analysis requirements consistent with the Technical Specification Surveillance Requirement (SR) 3.3.1.17. It is also consistent with the Core Operating Limits Report Requirement invoked by reference from Technical Specification Table 3.3.1-1, Notes 1 and 2. The two trip functions of interest are the "OPDT Reactor Trip" and the "OTDT Reactor Trip". This UFSAR Change does not degrade the RTD with respect to its qualification related to quality, environmental, seismic, design temperature, and pressure. The response time has been increased (made slower). The slower response time of the hot leg RTD's is enveloped by existing analyses.

The relevant design basis limit for fission product barriers in this case is fuel cladding limits (DNB and Linear Heat Generation Rates (kW/ft)). The response times for the RTDs are mainly in place for assessment of the fuel with respect to DNB limits and linear heat generation (kW/ft) limits for both fuel types (Mark-BW and RFA) contained in the Catawba Unit 1 Reactor. These limits are fundamental to barrier integrity, expressed numerically, and identified in the UFSAR. As such they qualify as design basis limits for fission product barriers. Since neither the DNB nor linear heat generation rates identified in the UFSAR are being changed, the response times in UFSAR Table 7-3 is assumed to be a subordinate parameter to the Design Basis Limits for Fission Product Barriers of DNB and kW/ft. Therefore no Design Basis Limits for Fission Product Barriers are being changed and thus none are being exceeded or altered.

A 10CFR50.59 evaluation concluded that this change could be made without prior NRC approval. No Technical Specification changes are required. UFSAR Table 7-3 will be revised.